NON-PUBLIC?: N
ACCESSION #: 9312010212
LICENSEE EVENT REPORT (LER)

FACILITY NAME: PLANT HATCH, UNIT 1 PAGE: 1 OF 8

DOCKET NUMBER: 05000321

TITLE: TRIP OF CONDENSATE PUMPS RESULTS IN DECREASING REACTOR WATER LEVEL AND A MANUAL SCRAM
EVENT DATE: 10/22/93 LER #: 93-013-00 REPORT DATE: 11/16/93

OTHER FACILITIES INVOLVED: PLANT HATCH UNIT 2 DOCKET NO: 05000366

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(i)
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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COMPLIANCE MANAGER, HATCH

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On 10/22/93 at 1744 CDT, Unit 1 was in the Run mode at a power level of 2436 CMWT (100% rated thermal power). At that time, the condensate pumps tripped. This caused a decrease in feedwater flow to the reactor vessel and a corresponding decrease in reactor water level. At 1744 CDT, with water level at approximately 18 inches above instrument zero and decreasing, personnel inserted a manual reactor scram. Water level decreased to a minimum of 52.4 inches below instrument zero (106 inches above the top of active fuel) due to the loss of feedwater flow and void collapse from the decrease in reactor power. A Group 2 Primary Containment Isolation System signal was received at about 12.3 inches above instrument zero. All Group 2 Primary Containment Isolation Valves (PCIVs) closed. Also, the recirculation system pumps tripped, the Unit 1 secondary containment isolated, the Unit 1 and Unit 2 Standby Gas

Treatment systems initiated, and the Group 5 PCIVs closed. The High Pressure Coolant Injection and Reactor Core Isolation Cooling systems automatically started when water level decreased to 35 inches below instrument zero. They restored reactor water level to greater than 12 inches above instrument zero within four minutes of the scram. All systems functioned per design. Thermal stratification led to violations of pressure and temperature and heatup rate limits. The trip of the condensate pumps appears to have been most likely caused by vibration in a condenser hotwell level instrument. This level instrument provides a trip signal to all the condensate pumps. The exact problem has not been determined. Corrective actions included trouble shooting the instrument and logic circuit and implementing a temporary modification to disable the pump trip on low hotwell level.

END OF ABSTRACT

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor
Energy Industry Identification System codes are identified in the text as (EIIS Code XX).

DESCRIPTION OF EVENT

On 10/22/93 at 1744 CDT, Unit 1 was in the Run mode at a power level of 2436 CMWT (100% rated thermal power). At that time, all three condensate pumps (EIIS Code SD), 1N21-C001A, B, and C, unexpectedly and simultaneously tripped. This resulted in a decrease in feedwater flow to the reactor vessel and a corresponding decrease in reactor water level from its normal level of 37 inches above instrument zero. At 1744 CDT, with reactor water level at approximately 18 inches above instrument zero (176 inches above the top of active fuel) and still decreasing, licensed operations personnel inserted a manual reactor scram in anticipation of an automatic scram on low reactor water level. All control rods fully inserted per design. Within 30 and 54 seconds of the trip of the condensate pumps, both reactor feedwater pumps (EIIS Code SJ) and all three condensate booster pumps (EIIS Code SD), respectively, tripped on low suction pressure per design. Reactor water level decreased to a minimum of 52.4 inches below instrument zero (106 inches above the top of active fuel) due to the loss of feedwater flow and void collapse from the decrease in reactor power.

As reactor water level decreased, an automatic scram signal and a Group 2 Primary Containment Isolation System (PCIS, EIIS Code JM) signal were received on low reactor water level at approximately 12.3 inches above instrument zero per design. All Group 2 Primary Containment Isolation Valves (PCIVs, EIIS Code JM) closed as required. The Reactor Recirculation system (EIIS Code AD) pumps tripped, the Unit 1 secondary containment isolated, and the Unit 1 and Unit 2 Standby Gas Treatment (EIIS Code BH) systems initiated at a reactor water level of 30 inches below instrument zero. The Group 5 PCIS received an isolation signal and the Group 5 PCIVs closed as water level decreased to 35 inches below instrument zero, isolating the Reactor Water Cleanup (EIIS Code CE) system. All systems functioned per design.

The High Pressure Coolant Injection (EIIS Code BJ) and Reactor Core Isolation Cooling (EIIS Code BN) systems automatically initiated and injected to the reactor vessel per design when reactor water level reached 35 inches below instrument zero. They, along with the inservice Control Rod Drive system (EIIS Code AA) pump, restored reactor water level to greater than 12 inches above instrument zero within four minutes of the manual scram. Operations personnel secured the High Pressure Coolant Injection system and maintained reactor water level greater than 12 inches above instrument zero using the Reactor Core Isolation Cooling and Control Rod Drive systems as necessary.

Reactor pressure was controlled by the Turbine Bypass Valves (EIIS Code SO). Pressure did not exceed its pre-event value of 990 psig. No Safety Relief Valves actuated nor were any required to actuate to reduce or control reactor pressure.

With both Reactor Recirculation system pumps tripped, forced circulation of the reactor coolant was no longer in effect. Relatively cold water was injected into the reactor vessel by the High Pressure Coolant Injection, Reactor Core Isolation Cooling, and Control Rod Drive systems to recover and maintain water level. As a result of these factors, the reactor coolant underwent thermal stratification resulting in rapidly decreasing water temperatures in the reactor vessel bottom head region. Within approximately 15 minutes of the scram, the reactor vessel bottom head drain line temperature had decreased to about 400 degrees Fahrenheit from its pre-scram value of 490 degrees Fahrenheit. At this point, the temperature difference between the reactor vessel steam dome and the bottom head drain was greater than 145 degrees Fahrenheit and per the requirements of Unit 1 Technical Specifications section 3.6.E, the Reactor Recirculation system pumps could not be re-started. Therefore, the thermal stratification could not be alleviated and the temperatures in the bottom head region continued to decrease.

By 2030 CDT on 10/22/93, bottom head drain line temperature had decreased to 206 degrees Fahrenheit with reactor pressure at 580 psig. This combination of drain line temperature and reactor pressure was outside the limits of Unit 1 Technical Specifications figure 3.6-2 and therefore not in compliance with Unit 1 Technical Specifications section 3.6.B.2. The bottom head drain line temperature continued to decrease due to thermal stratification and the injection of cold water from the Reactor Core Isolation Cooling and Control Rod Drive systems necessary to control reactor water level. As a result, the combination of drain line temperature and reactor pressure remained outside of the limits of Unit 1 Technical Specifications figure 3.6-2 until 1400 CDT on 10/23/93 when pressure had been decreased to 56 psig with drain line temperature at 78 degrees Fahrenheit, two degrees Fahrenheit above the limit for this pressure. Thereafter, pressures and temperatures remained within limits.

At approximately 0430 CDT on 10/23/93, the reactor vessel metal temperature in the area of the vessel bottom above the support skirt began to increase. The temperature increase was caused by the development of natural circulation in the reactor vessel resulting from the decreasing vessel pressure and an increase in water level. The temperature in the area of the vessel bottom head above the support skirt increased from 125 degrees Fahrenheit to 290 degrees Fahrenheit in one hour. This rate of temperature increase, 165 degrees Fahrenheit per hour, exceeded the limit of 100 degrees Fahrenheit per hour given in Unit 1 Technical Specification section 3.6.A. Following the initial increase to 290 degrees Fahrenheit, the temperature of the vessel bottom head above the support skirt remained fairly constant for the next seven hours, varying between 285 and 295 degrees Fahrenheit, before decreasing slowly over the following ten hours to 120 degrees Fahrenheit. No other temperature increased at a rate in excess of the Technical Specification limit.

CAUSE OF THE EVENT

The cause of the simultaneous trip of all three condensate pumps has not been determined conclusively. However, as discussed below, it is strongly suspected that the cause of the pump trip was a spurious actuation of the hotwell level instrument due to intermittent vibration. Low condenser hotwell (EIIS Code SQ) level will cause the simultaneous trip of the condensate pumps; however, there was no indication of a low hotwell level prior to the trip of the pumps. The chart recorder for the "B" condenser hotwell indicated a steady and normal hotwell level until the pumps tripped. It then showed a sudden increase in level as expected following the trip of the pumps. This indicated that the recorder was

functioning properly and following changes in hotwell level. A loop calibration of the level instrument and recorder performed after the event confirmed that the "B" condenser hotwell level instrument and recorder were in calibration and functioning properly. Therefore, this even
appears to have been caused by a problem in the "A" condenser hotwell level instrument, 1N21-R206. This level instrument provides a simultaneous trip signal to all three condensate pumps. No other single instrument or trip logic will cause all of the condensate pumps to trip at the same time. The exact problem in the instrument, however, has not been determined.

Condenser hotwell level instrument 1N21-R206 was calibrated following this event; the instrument was found to be in calibration with the low level annunciation and trip setpoints set correctly. No problems with the operation of the instrument were noted during the calibration.

The instrument's level switch actuator housing was examined. It was found to be clean and dry. No problems were found which would have resulted in the generation of a low level trip signal to the condensate pumps. One wire with worn insulation was found in the housing; however, the wire was part of the low level annunciation circuit. The annunciation circuit is separate from the trip logic circuit; thus, this problem could not have caused the low level trip. The wire was repaired.

Also, the low level trip relay coils and the associated wiring in the trip logic circuit were checked for grounding problems. No evidence of grounds which might have resulted in a trip of the condensate pumps was found.

Further investigations have found that trips of the hotwell low level annunciator circuit are occurring concurrent with a recently discovered intermittent banging noise and vibration in the main condenser. Apparently, the vibration is causing the annunciator contacts to pick up; however, energization of the switch which causes a trip of the condensate pumps has not been seen. Voltage dips, however, have been observed. It cannot be conclusively determined, therefore, that the condensate pump trip contacts are energizing coincident with the banging noise in the main condenser. Nevertheless, this appears to be the most likely cause of the trip at this time. The source of the noises in the condenser is still under investigation.

The cause of operation outside of the limits of Unit 1 Technical Specifications sections 3.6.A and 3.6.B.2 was thermal stratification of the reactor coolant. The thermal stratification occurred upon loss of forced circulation in the vessel, when the Reactor Recirculation system pumps tripped on low reactor water level, in conjunction with the injection of relatively cold water necessary to restore and maintain water level. These factors caused vessel bottom head area temperatures to decrease preventing re-start of the Reactor Recirculation system pumps and resulting in continuing thermal stratification of the reactor coolant. Thermal stratification, in turn, led to a large increase in a vessel bottom area temperature when natural circulation caused hotter water from the annulus region to flow past the colder vessel support skirt area. Additionally, it led to a decrease in vessel bottom head drain line temperature such that pressure and temperature limits were not met.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required by 10 CFR 50.73(a)(2)(iv) because of the unplanned actuation of Engineered Safety Feature systems. Specifically, the simultaneous trip of all three condensate pumps resulted in an unplanned manual actuation of the Reactor Protection System (EIIS Code JC) as a result of decreasing reactor water level. The manual scram was followed by an automatic scram signal and a Group 2 PCIS isolation signal on low reactor water level. The Unit 1 secondary containment isolated, the Unit 1 and Unit 2 Standby Gas Treatment systems actuated, the Group 5 PCIVs closed on a Group 5 PCIS isolation signal, and the High Pressure Coolant Injection system automatically started and injected to the reactor vessel on low low reactor water level per design.

This report also is required by 10 CFR 50.73(a)(2)(i) because a condition existed that was prohibited by the plant's Technical Specifications. Specifically, the temperature in the reactor pressure vessel bottom head drain line was outside the pressure and temperature limits of Unit 1 Technical Specifications figure 3.6-2. In addition, the heatup rate in the area of the vessel bottom head above the support skirt exceeded the limit of 100 degrees Fahrenheit per hour given in Unit 1 Technical Specifications section 3.6.A.

The Condensate/Feedwater system consists, in part, of three condensate pumps, three condensate booster pumps, and two reactor feedwater pumps. The condensate pumps take a suction from the "A" and "B" condenser hotwells through a common header which is connected to both hotwells. The condensate booster pumps take a suction from the discharge of the condensate pumps, and the feedwater pumps, in turn, take a suction from the discharge of the condensate booster pumps. The feedwater pumps discharge to the reactor vessel. In order to protect the pumps from damage due to cavitation, each pump has a protective trip on low suction pressure. In the case of the condensate pumps, this protective trip takes the form of an immediate trip on low condenser hotwell level since this level determines the condensate pumps' suction pressure. (The condensate booster pumps and feedwater pumps have time delays on their low suction pressure trips to prevent unnecessary trips during short lived system transients.) Because of the arrangement of the pumps, a trip of two or more of the same type of pump

(i.e., condensate pump or condensate booster pump) will almost always result in a trip of the downstream pumps on low suction pressure when the unit is operating at 100% rated thermal power.

In this event, all three condensate pumps tripped. Feedwater flow and, consequently, reactor water level decreased. Licensed operations personnel manually scrammed the reactor in anticipation of an automatic scram on low reactor water level. As reactor water level decreased due to the loss of feedwater flow and void collapse from the decrease in reactor power, various Engineered Safety Feature systems actuated per design. The High Pressure Coolant Injection and Reactor Core Isolation Cooling systems automatically initiated and injected to the reactor vessel when water level dropped to 35 inches below instrument zero. Water level was restored to greater than 12 inches above instrument zero within four minutes of the manual scram. At no time was level less than 106 inches above the top of active fuel. Reactor pressure was controlled with the Turbine Bypass Valves and never exceeded its pre-event value of 990 psig. All systems functioned per design given the unexpected trip of the condensate pumps.

Unit 1 Technical Specifications figure 3.6-2 provides the reactor vessel pressure and temperature limits for non-nuclear heatup and cooldown as well as for low power physics testing. These limits are based on the fracture toughness analysis of the reactor vessel for these conditions. A heatup and cooldown rate of 100 degrees Fahrenheit was assumed in developing the pressure and temperature limits given in figure 3.6-2.

The figure reflects the pressure and temperature limits for the core beltline material. They are more conservative than the pressure and temperature limits for other areas of the reactor vessel, such as the bottom head region, because of the embrittlement of the material in the beltline region resulting from fast neutron exposure. The fracture toughness of ferritic steels gradually decreases with increasing fast neutron exposure; therefore, the fracture toughness of the material in the core beltline region will be the most limiting since it is the area of the vessel which is exposed to fast neutrons.

In this event, the temperature in the bottom head drain line was outside the pressure and temperature limits of figure 3.6-2. In addition, the heatup rate in the area of the reactor vessel bottom head above the support skirt exceeded the limit of 100 degrees Fahrenheit per hour. These two conditions were analyzed. It was determined that no functional or structural integrity concerns existed regarding operation outside of the pressure and temperature limits. Also, the effects on vessel stress and fatigue due to the heatup event were determined to be small, with adequate margin remaining to the limits given in the applicable section of the American Society of Mechanical Engineer's Boiler and Pressure Vessel Code.

Based on the above analysis, it is concluded that this event had no adverse impact on nuclear safety. This analysis is applicable to all power levels.

CORRECTIVE ACTIONS

The "A" condenser hotwell level instrument, 1N21-R206, and the condensate pump trip logic circuit were checked for problems as described previously. No problems were found which would have caused the simultaneous trip of all the condensate pumps.

A temporary modification was implemented to disable the Unit 1 condensate pump trip on low condenser hotwell level. The permanent deletion of the low level pump trip is being considered.

The situation concerning the vibration of the 1N21-R206 switches, concurrent with the main condenser banging noises is still being investigated. At the present time, the source of the condenser noises is also under investigation.

It was concluded, prior to unit startup, that there were no functional or structural integrity concerns as a result of operation outside of the pressure and temperature limits. It also was concluded that the effects on vessel stress and fatigue due to the heatup event were small, with adequate margin remaining to the limits given in the applicable section of the Boiler and Pressure Vessel Code.

Current plans for prevention of future thermal stratification events include submittal of a request to revise Technical Specifications. A change will be requested to allow restart of the Reactor Recirculation pumps without regard to differential temperatures provided certain restrictions are met which ensure acceptable vessel stresses. A change to the allowable value for the low reactor vessel level Anticipated Transient Without Scram Recirculation Pump Trip setpoint will also be requested to prevent the Reactor Recirculation pumps from tripping due to a loss of feedwater event. General Electric has been involved in the analyses of the event, and in support of the Technical Specification changes. They are, therefore, aware of the potential generic implications of thermal stratification issues.

ADDITIONAL INFORMATION

No systems other than those mentioned in this report were affected by this event.

No failed components could be found which caused or contributed to this event.

Similar events in the last two years in which a loss of feedwater flow resulted in a manual or automatic reactor scram have been reported in Licensee Event Reports 50-321/1992-009, dated 4/23/92, and 50-366/1992-009, dated 7/24/92. Both of these events were caused by personnel error and corrective actions for the events were directed toward prevention of similar personnel errors. The event described in this Licensee Event Report was not caused by personnel error; it appears to have been caused instead by a mechanical or electrical failure. Therefore, corrective actions addressing personnel error could not h
ve
prevented this event.

A similar event in the last two years in which the pressure and temperature and the heatup limits were exceeded was reported in Licensee Event Report 50-321/1992-023, Revision 1, dated 12/23/92.

ATTACHMENT 1 TO 9312010212 PAGE 1 OF 1

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J. T. Beckham, Jr. Georgia Power
Vice President - Nuclear
Hatch Project the southern electric system

November 16, 1993
Docket No. 50-321 HL-4430

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Edwin I. Hatch Nuclear Plant - Unit 1
Licensee Event Report
Trip of Condensate Pumps Results in a Manual Scram

Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(i) and (iv), Georgia Power Company is submitting the enclosed Licensee Event Report (LER) concerning a trip of the condensate pumps which resulted in decreasing reactor pressure vessel water level and a manual scram. This event occurred at Plant Hatch Unit 1.

Sincerely,

J. T. Beckham, Jr.

OCV/cr

Enclosure: LER 50-321/1993-013

cc: Georgia Power Company
Mr. H. L. Summer, General Manager - Nuclear Plant
NORMS

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. K. Jabbour, Licensing Project Manager - Hatch

U.S. Nuclear Regulatory Commission. Region II
Mr. S. D. Ebneter, Regional Administrator
Mr. L. D. Wert, Senior Resident Inspector - Hatch

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